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SCDAP/RELAP5-3D© CODE MANUAL

VOLUME 1: CODE ARCHITECTURE AND INTERFACE OF THERMAL HYDRAULIC AND CORE BEHAVIOR MODELS

SCDAP/RELAP5-3D© Code Development Team

BECHTEL BWXT IDAHO, LLC

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ABSTRACT

The SCDAP/RELAP5-3D© code has been developed for best-estimate transient simulation of light water reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system and reactor core during severe accidents as well as large and small break loss-of-coolant accidents, operational transients such as anticipated transient without SCRAM, loss of offsite power, loss of feedwater, and loss of flow. The coolant system behavior is calculated using a two-phase model allowing for unequal temperatures and velocities of the two phases of the fluid, and the flow of fluid through porous debris and around blockages caused by reactor core damage. The reactor core behavior is calculated using models for the ballooning and oxidation of fuel rods, the meltdown of fuel rods and control rods, fission product release, and debris formation. The code also calculates the heatup and structural damage of the lower head of the reactor vessel resulting from the slumping of reactor core material. A generic modeling approach is used that permits as much of a particular system to be modeled as necessary. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems.

This volume describes the organization and manner of the interface between severe accident models in the SCDAP portion of the code and hydrodynamic models in the RELAP5 portion of the code. A description is also given of the overall architecture of the code. Additional information is provided regarding the manner in which models in one portion of the code impact other parts of the code, and models which are dependent on and derive information from other subcodes.

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EXECUTIVE SUMMARY

The specific features of SCDAP/RELAP5-3D© are described in this five volume set of manuals covering the theory, use, and assessment of the code for severe accident applications.

The SCDAP/RELAP5-3D© computer code is designed to calculate for severe accident situations the overall reactor coolant system (RCS) thermal-hydraulic response, core damage progression, and reactor vessel heatup and damage. The SCDAP/RELAP5-3D© code evolved from the RELAP5 and SCDAP/RELAP5 codes developed at the Idaho National Engineering & Environmental Laboratory (INEEL) under sponsership by the U.S. Nuclear Regulatory Commission (US NRC). Development of the RELAP5 code series began at the INEEL in 1975, while SCDAP development was initiated in the early 1970's with an active linkage to RELAP5 in 1979. The SCDAP/RELAP5-3D© code maintained all of the capabilities and validation history of the predecessor codes, plus the added capabilities sponsored by the DOE.

The RELAP5 code is based on a two-fluid model allowing for unequal temperatures and velocities of the fluids and the flow of fluid through porous debris and around blockages caused by reactor core damage. The models in SCDAP calculate the progression of damage to the reactor core. These models calculate the heatup, oxidation and meltdown of fuel rods and control rods, the ballooning and rupture of fuel rod cladding, the release of fission products from fuel rods, and the disintegration of fuel rods into porous debris and molten material. The SCDAP models also calculate the heatup and structural damage of the reactor vessel lower head resulting from the slumping to the lower head of reactor core material with internal heat generation. SCDAP/RELAP5-3D© can be used in analyses of fission product transport and deposition behavior and containment phenomena by linking it to the detailed fission product code, VICTORIA[a] or CONTAIN[b], respectively.

The SCDAP/RELAP5-3D© code includes many generic component models from which general systems can be simulated. The component models include fuel rods, control rods, pumps, valves, pipes, reactor vessel, electrical fuel rod simulators, jet pumps, turbines, separators, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and noncondensable gas transport. The code also includes a model for reactor kinetics.

This volume, Volume 1, describes the organization and manner of interface between the severe accident models in the SCDAP portion of the code and the hydrodynamic models in the RELAP5 portion of the code. A description is also given of the overall architecture of the code.

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a. N. E. Bixler, "VICTORIA2.0: A Mechanistic model for Radionuclide Behavior in a Nuclear Reactor Coolant System Under Severe Accident Conditions," NUREG/CR-6131, SAND93-2301, December 1998.
b. K. D. Bergeron et al., *User's Manual for CONTAIN 1.0, A Computer Code for Severe Nuclear Reactor Accident Containment Analysis*, NUREG/CR-4085, SAND84-1204, May 1985

1. INTRODUCTION

The SCDAP/RELAP5-3D© computer code is designed to calculate for severe accident situations the overall reactor coolant system (RCS) thermal-hydraulic response, reactor core and vessel damage progression, and, in combination with VICTORIA,[1] fission product release and transport during severe accidents.

1.1 General Code Capabilities

The SCDAP/RELAP5-3D© code contains RELAP5 and SCDAP models. The RELAP5 models calculate the overall RCS thermal-hydraulics, control system interactions, reactor kinetics, and transport of noncondensable gases. A model is also included in RELAP5 to calculate flow losses in porous debris. Although previous versions of the code have included the analysis of fission product transport and deposition behavior using models derived from TRAP-MELT, this capability has been replaced through a data link to the detailed fission product code, VICTORIA. The SCDAP models calculate the heatup and damage progression in the core structures and the lower head of the reactor vessel. The calculations of damage progression include calculations of the meltdown of fuel rods and structures, the fragmentation of embrittled fuel rods, convective and radiative heat transfer in porous debris, the formation of a molten pool of core material, and the slumping of molten material to the lower head.

SCDAP/RELAP5-3D© is capable of modeling a wide range of system configurations from single pipes to different experimental facilities to full-scale reactor systems. The configurations can be modeled using an arbitrary number of fluid control volumes and connecting junctions, heat structures, core components, and system components. Flow areas, volumes, and flow resistances can vary with time through either user-control or models that describe the changes in geometry associated with damage in the core. System structures can be modeled with RELAP5 heat structures, SCDAP core components, or SCDAP debris models. The RELAP5 heat structures are one-dimensional models with slab, cylindrical, or spherical geometries. The SCDAP core components include representative light water reactor (LWR) fuel rods, silver-indium-cadmium (Ag-In-Cd) and B_4C control rods and/or blades, electrically heated fuel rod simulators, and general structures. A two-dimensional, finite element heat conduction model based on the COUPLE[2] code may be used to calculate the heatup of the lower head of the reactor vessel and the slumped material supported by the lower head. This model takes into account the decay heat and internal energy of newly fallen or formed debris and then calculates the transport by conduction of this heat in the radial and axial directions to the wall structures and water surrounding the debris. The most important use of this model is to calculate the heatup of the vessel lower head and the timing of its failure in response to contact with material that has slumped from the core region. Other system components available to the user include pumps, valves, electric heaters, jet pumps, turbines, separators, and accumulators. Models to describe selected processes, such as reactor kinetics, control system response, and tracking noncondensable gases, can be invoked through user control.

The SCDAP/RELAP5-3D© code evolved from the RELAP5 and SCDAP/RELAP5 codes developed at the Idaho National Engineering & Environmental Laboratory (INEEL) under sponsership by the U.S.

Nuclear Regulatory Commission (US NRC). Development of the RELAP5 code series began at the INEEL in 1975, while SCDAP development was initiated in the early 1970's with an active linkage to RELAP5 in 1979. Following the accident at Chernobyl, the U.S. Department of Energy (DOE) began a re-assessment of the safety of its test and production reactors, and chose RELAP5 and SCDAP/RELAP5 as the analytical tools for system safety analysis because of their wide spread acceptance and ease of application to such widely varying systems. Systematic safety analyses were performed for the N reactor at Hanford, the K and L reactors at Savannah River, the Advanced Test Reactor (ATR) at INEEL, the High Flux Isotope Reactor (HFIR) and Advanced Neutron Source (ANS) at Oak Ridge, and the High Flux Beam Reactor at Brookhaven. DOE also chose RELAP5 for the independent safety analysis of the New Production Reactor (NPR) before that program was cancelled.

The application of SCDAP/RELAP5 and RELAP5 to these widely varying reactor designs demanded new modeling capabilities, including non-light water reactor (LWR) materials and geometry. These widely varying demands were met by maintaining a single source with options that could be selected or deselected at compilation. In this fashion both NRC and DOE users could receive maximum benefit from the others development efforts. After the transmittal of SCDAP/RELAP5 MOD3.3 to the NRC, it became clear, however, that the efficiencies realized by the maintenance of a single source code for use by both NRC and DOE were being overcome by the extra effort required to accommodate sometimes conflicting goals and requirements. The codes were therefore "split" into two versions, SCDAP/RELAP5 MOD3.3 for the NRC and SCDAP/RELAP5-3D© for DOE. The SCDAP/RELAP5-3D© code maintained all of the capabilities and validation history of the predecessor codes, plus the added capabilities sponsored by the DOE.

SCDAP/RELAP5-3D© is the latest INEEL-developed code for analyzing transients and accidents in water-cooled nuclear power plants and related systems. The most prominent attribute that distinguishes this code from its predecessors is the fully integrated, multi-dimensional thermal-hydraulic and kinetic modeling capability. Although multi-dimensional capabilities in the RELAP models have been assessed, it should be noted that few of these assessment calculations have used SCDAP models.

1.2 Relationship to Other Software

SCDAP/RELAP5-3D© and RELAP5-3D© were developed in parallel and share a common configuration. Both codes share a common source deck. Separate codes are formed only prior to compilation, so changes made to the source deck are automatically reflected in both codes.

The development and application of the code is also related to several other oftware packages. Theoretical work associated with the development of PARAGRASS-VFP[3] has resulted in model improvements for fission product release. A link with PATRAN[4] and ABAQUS[5] provides the user with the means to calculate the details of lower head failure. Animated plant response displays are possible through links to the Nuclear Plant Analyzer (NPA)[6] display software, which gives the user an efficient way of analyzing the large amount of data generated. Detailed plant simulations from accident initiation through release of fission products to the atmosphere are possible by applying SCDAP/RELAP5-3D©

results to analyses with VICTORIA[1] code for fission product release and transport, CONTAIN[7] code for containment response and CRAC2[8] or MACCS[9] codes for atmospheric dispersion consequence.

1.3 Quality Assurance

SCDAP/RELAP5-3D© is maintained under a strict code configuration system that provides a historical record of the changes made to the code. Changes are made using an update processor that allows separate identification of improvements made to each successive version of the code. Modifications and improvements to the coding are reviewed and checked as part of a formal quality program for software. In addition, the theory and implementation of code improvements are validated through assessment calculations that compare the code-predicted results to idealized test cases or experimental results.

1.4 Organization of the SCDAP/RELAP5-3D© Manuals

The specific features of SCDAP/RELAP5-3D© are described in a five-volume set of manuals covering the theory (Volume 2), user's guidelines and input manual (Volume 3), material properties (Volume 4), and assessment (Volume 5). Although Volume 1 describes (a) the overall code architecture, (b) interfaces between the RELAP5 and SCDAP models, and (c) any system models unique to SCDAP/RELAP5-3D©, the code user is referred to the companion set of six volumes which describe the RELAP5-3D© system thermal-hydraulics and associated models.

Volume 1 presents a description of SCDAP/RELAP5-3D©-specific thermal-hydraulic models (relative to RELAP5-3D©), and interfaces between the thermal-hydraulic models and damage progression models. Volume 2 contains detailed descriptions of the severe accident models and correlations. It provides the user with the underlying assumptions and simplifications used to generate and implement the basic equations into the code, so an intelligent assessment of the applicability and accuracy of the resulting calculation can be made. Volume 3 provides the user's guide and code input for the severe accident modeling. User guidelines are produced specifically for the severe accident code. The user should also refer to the RELAP5-3D© Code Manual Volume V: User Guidelines for a complete set of guidelines. Volume 4 describes the material property library, MATPRO. It contains descriptions of the material property subroutines available for severe accident analysis. Volume 5 documents code assessments. It summarizes the improvements made to various versions of the code and the effect of these improvements on code calculations. A presentation is made of the comparisons of code calculations of a wide range of severe fuel damage experiments with the measured results of these experiments. Also presented are code calculations of the TMI-2 accident and calculations of severe accidents in typical PWRs and BWRs.

1.5 Organization of Volume I

Volume 1 describes the architecture of SCDAP/RELAP5-3D© and the interface of the thermal hydraulic models with the reactor core and vessel models. The organization and structure of SCDAP/RELAP5-3D© is described in Section 2 of the volume. Section 3 describes the manner in which models in one portion of the code impact other parts of the code. Section 4 describes extensions to RELAP5-3D© models and to its procedure for time step control. Section 5 provides references.

2. CODE ARCHITECTURE

Modeling flexibility, user convenience, and computer efficiency were primary considerations in the development of SCDAP/RELAP5-3D©. The following sections describe computer adaptability, code top level organization, input processing, and transient operation.

2.1 Computer Adaptability

2.1.1 Source Coding

SCDAP/RELAP5-3D© was originally written in FORTRAN 77 but now includes many FORTRAN 90 features. Compile time-options are provided to allow operation on 64-bit machines and 32-bit machines that have double-precision (64-bit), floating-point arithmetic. A common source is maintained for all computer versions. Reported errors are resolved on all computer versions derived from the common source.

SCDAP/RELAP5-3D© minimizes the need for hardware specific coding through the use of generic functions, character variables and statements, and open statements. The bit handling functions used are from a Mil-spec standard and many computers have implemented that standard. Some machine dependent coding is needed to overcome deficiencies in the Fortran standard. For example, the standard does not allow specification of variable range and precision requirements. Unless modified, the Fortran for a 64-bit machine in single precision would use 64 bits while the 32-bit machine would use only 32 bits. Additional statements are needed to indicate double precision on the 32-bit machine. Some compilers have options for automatically converting to double precision, but that is nonstandard and not available on all systems. Nonstandard coding is needed in some subroutines to define the smallest and largest floating point numbers, the smallest floating point increment, and to access the radix, fraction, and power part of a floating point number. The next standard should remedy these, but until then, precompilers are used to handle hardware and software differences.

2.1.2 Systems

The SCDAP/RELAP5-3D© computer program should execute on a wide variety of scientific computers with minimal modifications. In particular, the code should execute on all 64-bit computers, that is computers using 64 bits for both floating point and integer arithmetic. It should also execute on the multitude of 32-bit computers that range from workstations to supercomputers and that have 32-bit integer arithmetic but provide 64-bit floating point arithmetic through double precision operations. The code is maintained for all computers in a common source file, and through one or two stages of precompiling, the code is made suitable for a particular computer. Table 2-1 lists the computers where SCDAP/RELAP5-3D© and RELAP5-3D© test problems have been run.

Table 2-1. Computers executing SCDAP/RELAP5-3D© or RELAP5-3D©.

Computer Identification
Cray X-MP 2/16
DEC Alpha 3000/500X
DEC 5000/200
HP 720
HP 735
HP 750
IBM R6000/320
IBM R6000/370
IBM R6000/540
IBM R6000/Power2/590
SGI Indigo
SGI Crimson
Sun Sparc2
Sun Sparc
Windows-based PCs

2.2 Organization of the Code

SCDAP/RELAP5-3D© is coded in a modular fashion using top-down structuring. The various models and procedures are isolated in separate subroutines. Figure 2-1 shows an overview of the code architecture.

Input processing is performed in INPUTD and associated subroutines. Transient control is performed by TRNCTL and associated subroutines. The STRIPF routine extracts data from the restart plot file for use in other computer programs. Because of their complexity, the input processing and transient control routines are described in more detail in the Sections 2.2.1 and 2.2.2.

2.2.1 Input Processing Overview (INPUTD and RNEWP)

The input processing is performed in three phases. In the first phase, the input data is read and checks are made for typing and punctuation errors (such as multiple decimal points and letters in numerical fields), and stores the data keyed by card number so that the data are easily retrieved. A listing of the input data is provided, and punctuation errors are noted.

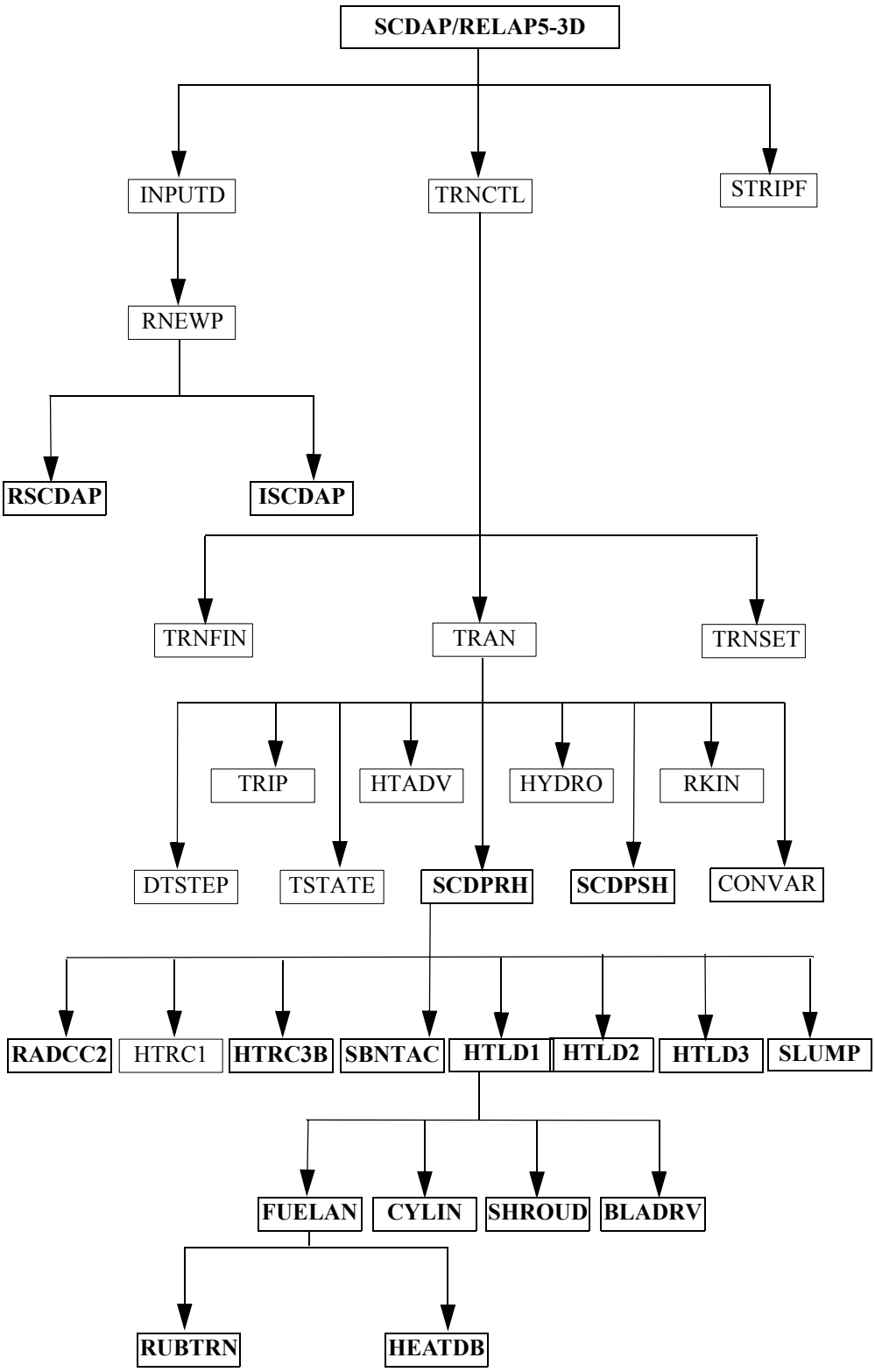

Figure 2-1. SCDAP/RELAP5-3D© architecture with SCDAP specific routines highlighted.

During the second phase, restart data from a previous simulation are read if the problem is a RESTART type, and all input data are processed. Some (common) processed input is stored in fixed common blocks, but the majority of the data are stored in dynamic data blocks that are created only if needed by a problem and sized to the particular problem. In a NEW-type problem, dynamic blocks must be created. In RESTART problems, dynamic blocks may be created, deleted, added to, partially deleted, or modified as modeling features and components within models are added, deleted, or modified. Extensive input checking is done, but at this level checking is limited to new data from the cards being processed. Relationships with other data cannot be checked because the latter may not yet be processed. As an illustration of this level of checking, junction data are checked to determine if they are within the appropriate range, such as positive, nonzero, or between zero and one; and volume connection codes are checked for proper format. However, no attempt is made at this point to check whether or not referenced volumes exist in the problem until all input data are processed.

The third phase of processing begins after all input data have been processed. Because all data have been placed in fixed common or dynamic data (common) blocks during the second phase, complete checking of interrelationships can proceed. Examples of cross-checking are existence of hydrodynamic volumes referenced in junctions and heat structure boundary conditions; entry or existence of material property data specified in heat structures; and validity of variables selected for minor edits, plotting, or used in trips and control systems. As the cross-checking proceeds, cross-linking of the data blocks is done so that it need not be repeated at every time step. The initialization required to prepare the model for start of transient advancement is done at this level.

Input data editing and diagnostic messages can be generated during the second and/or third phases. Input processing for most models generates output and diagnostic messages during both phases.

As errors are detected, various recovery procedures are used so that input processing can be continued and a maximum amount of diagnostic information can be furnished. Recovery procedures include supplying default or replacement data, marking the data as erroneous so that other models do not attempt use of the data, or deleting the bad data. The recovery procedures sometimes generate additional diagnostic messages. Often after attempted correction of input, different diagnostic messages appear. These can be due to continued incorrect preparation of data, but the diagnostics may result from the more extensive testing permitted as previous errors are eliminated.

The input processing for the SCDAP portion of the code is performed in two main subroutines as shown in Figure 2-1, RSCDAP and ISCDAP. These subroutines perform the following functions:

RSCDAP Processes the input based upon the RELAP5 approach.

ISCDAP Initializes the SCDAP related variables in the code, maps fuel element locations into thermal-hydraulic volumes, and performs input consistency checks once all input data has been read in.

2.2.2 Transient Overview (TRNCTL)

Subroutine TRNCTL consists only of the logic to call the next lower level routines.

Subroutine TRNSET brings dynamic blocks required for transient execution from disk into computer central memory, performs final cross-linking of information between data blocks, sets up arrays to control the sparse matrix solution, establishes scratch work space, and returns unneeded computer memory. Subroutine TRAN, the driver, controls the transient advancement of the solution. Nearly all the execution time is spent in this subroutine, and TRAN is also the most demanding of memory. The subroutine TRNFIN releases space for the dynamic data blocks that are no longer needed and prints the transient timing summary.

The following description is presented for selected subroutines driven by TRAN (see Figure 2-1):

DTSTEP Determines the time step size, controls output editing, and determines whether transient advancements should be terminated. During program execution, this module displays such information as CPU time, problem time, and the maximum cladding temperature on a terminal screen.

TRIP Evaluates logical statements. Each trip statement is a simple logical statement which has a true or false result. The decision of what action is needed resides within the components in other modules. For example, valve components open or close the valve based on trip values; pump components test trip status to determine whether a pump electrical breaker has tripped.

TSTATE Calculates the thermodynamic state of the fluid in each hydrodynamic user-defined time-dependent volume.

HTADV Advances heat conduction/transfer solutions using previous-time-step reactor kinetics power and previous-time-step hydrodynamic conditions for computing heat transfer coefficients. It calculates heat transferred across solid boundaries of hydrodynamic volumes.

SCDPRH Advances the heat conduction, mechanical response (including changes in geometry), and fission gas release models using previous-time-step hydrodynamic conditions. It is in this block that nearly all of the SCDAP core component routines are exercised.

HYDRO Advances the hydrodynamic solution.

SCDPSH Drives the COUPLE subcode.

RKIN Advances the reactor kinetics of the code. It computes the power behavior in a nuclear reactor using the space-independent or point kinetics approximation which assumes that power can be separated into space and time functions.

CONVAR Provides the capability of simulating control systems typically used in hydrodynamic systems. It consists of several types of control components. Each component defines a control variable as a specific function of time-advanced quantities. The time-advanced quantities include quantities from hydrodynamic volumes, junctions, pumps, valves, heat structures, reactor kinetics, trip quantities, and the control variables themselves. This permits control variables to be developed from components that perform simple, basic operations.

RADCC2 Calculates the radiation heat transfer in a fuel bundle.

HTRC1 Computes heat transfer coefficients for air-water mixtures, single-phase liquids, subcooled nucleate boiling, saturated nucleate boiling, subcooled transition film boiling, saturated transition film boiling, subcooled film boiling, saturated film boiling, and single-phase

vapor convection.

HTRC3B Calculates convective and radiative porous heat transfer from debris to coolant.

SBNTAC Drives all SCDAP components.

HTLD1[a] Calculate the formation and heatup of a molten pool in the core region. calculate the spreading of the molten pool.

SLUMP Determines whether a new unique slumping of core material into lower vessel region occurred during a time step. If slumping occurred, it calculates the total mass of material that will end up eventually falling into the lower vessel region due to this slumping. This falling may be spread out over many time steps.

FUELAN Calculates behavior of LWR fuel rod component.

CYLIN Calculates behavior of LWR control rod component.

SHROUD Calculates behavior of shroud component.

BLADRV Calculates behavior of the control blade/channel box component.

RUBTRN Identifies locations that have degenerated into debris and calculates characteristics of debris.

HEATDB Calculates heatup of porous debris in core region.

Although there is a conceptual boundary between the hydrodynamic and severe core damage subroutines of the SCDAP/RELAP5-3D© code, each is intimately bound to the other and information is freely exchanged across the conceptual boundary.

a. (also HTLD2 and HTLD3)

3. INTERFACE WITH RELAP5 MODELS

This section describes the interface of SCDAP and its models for severe core damage with RELAP5 and its models for thermal-hydraulic behavior. The exchange of information between these two parts of the code are also summarized.

3.1 Common Data

The exchanges of information between SCDAP and RELAP5 occur through the medium of common blocks. The RELAP5 common blocks named /voldat/ and /jundat/ contain the variables that are used and modified by SCDAP. These RELAP5 variables are used and modified in several different subroutines in SCDAP. The SCDAP subroutine named SCDPRH is called by the RELAP5 subroutine named TRAN to account for the behavior of SCDAP heat structures on the behavior of the fluid in the region of the reactor core and lower head of the reactor vessel.

3.2 Variable Exchanges Between SCDAP and RELAP5

This section describes the flow of information between the SCDAP and RELAP5 parts of the SCDAP/RELAP5-3D© code. This flow of information occurs at every time step and results in an active link of the SCDAP models with the RELAP5 models. The variables in the flow of information calculated in SCDAP and then passed to RELAP5 are listed and a description is given of the impact on the RELAP5 calculations of these variables received from SCDAP. Then, the variables calculated in RELAP5 and passed to SCDAP are listed and a description is given of the impact on the SCDAP calculations of the variables received from RELAP5. A summary of the flow of information from SCDAP to RELAP5 is shown in Figure 3-1. This figure identifies each subroutine in SCDAP that supplies information to RELAP5 and each subroutine in RELAP5 that applies the information received from SCDAP. The names of the variables that contain the transmitted information are also shown. The flow of information from RELAP5 to SCDAP is shown in Figure 3-2.

The variables calculated by SCDAP and passed on to RELAP5 are: (1) surface temperatures of intact and debris SCDAP heat structures and COUPLE heat structures, (2) heat transferred to fluid by radiation from intact heat structures, (3) changes in flow area and hydraulic diameter, (4) rate of consumption of steam and the corresponding production of hydrogen, (5) release of fission gases such as Xe and Kr, and (6) heat transferred from porous debris to fluid in contact with the porous debris, heat transferred from a molten pool to fluid, and heat transferred from slumping core material to fluid. These variables and their impact on RELAP5 calculations are described next.

The surface temperatures of SCDAP and COUPLE heat structures are passed to RELAP5 in order to calculate the convective heat transfer at the surface of the heat structures. These variables are named tcond3(is,j,k) and tz(js), where “is” is the radial node at the surface of the SCDAP heat structures, “j” is the axial node number, “k” is the core component identification number, and “js” is a COUPLE heat structure node number located at the surface. These variables impact the calculated temperature and vapor generation rate for the fluid that interfaces with SCDAP and COUPLE heat structures. The surface

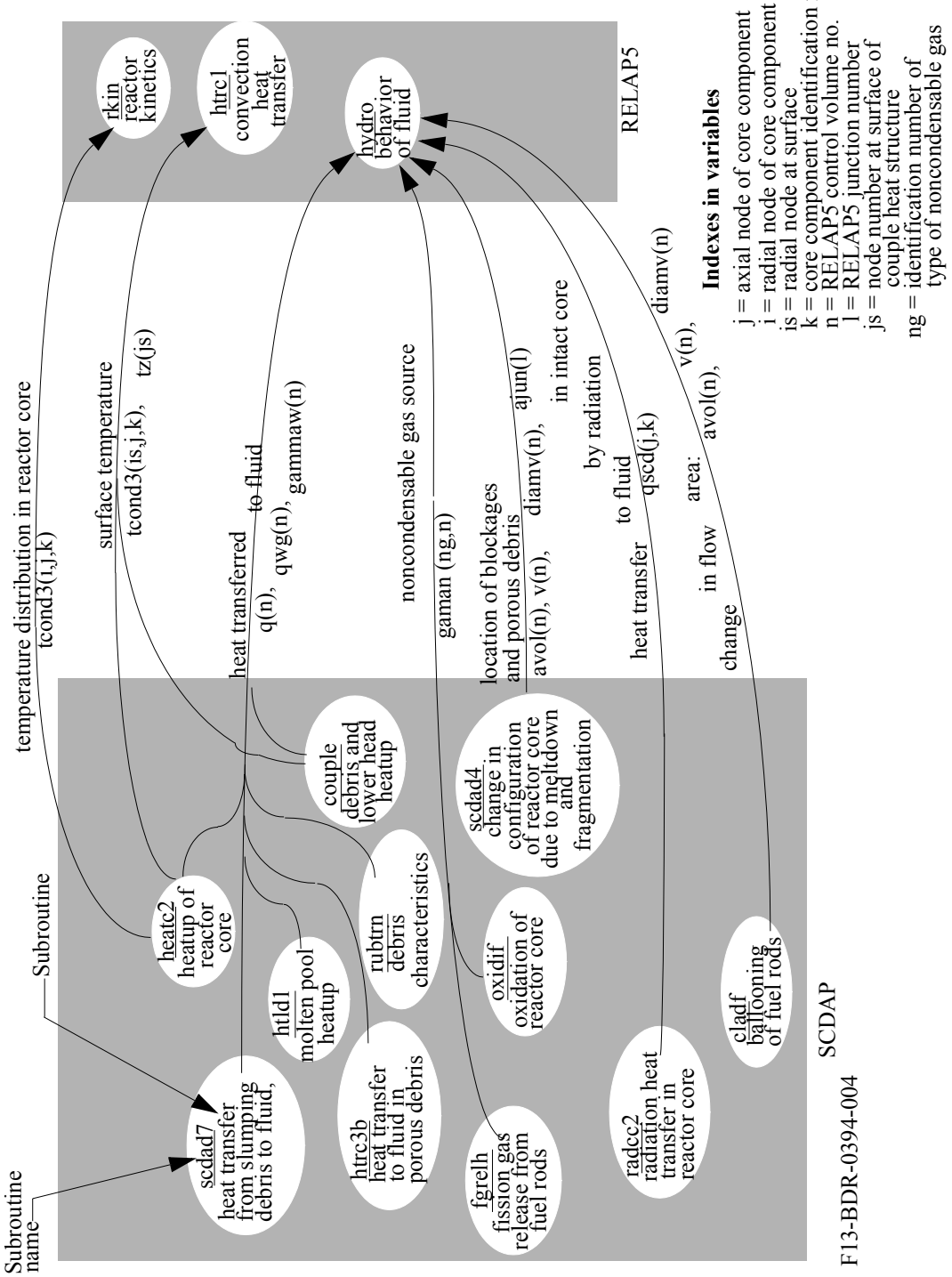

Figure 3-1. Flow of information from SCDAP to RELAP5.

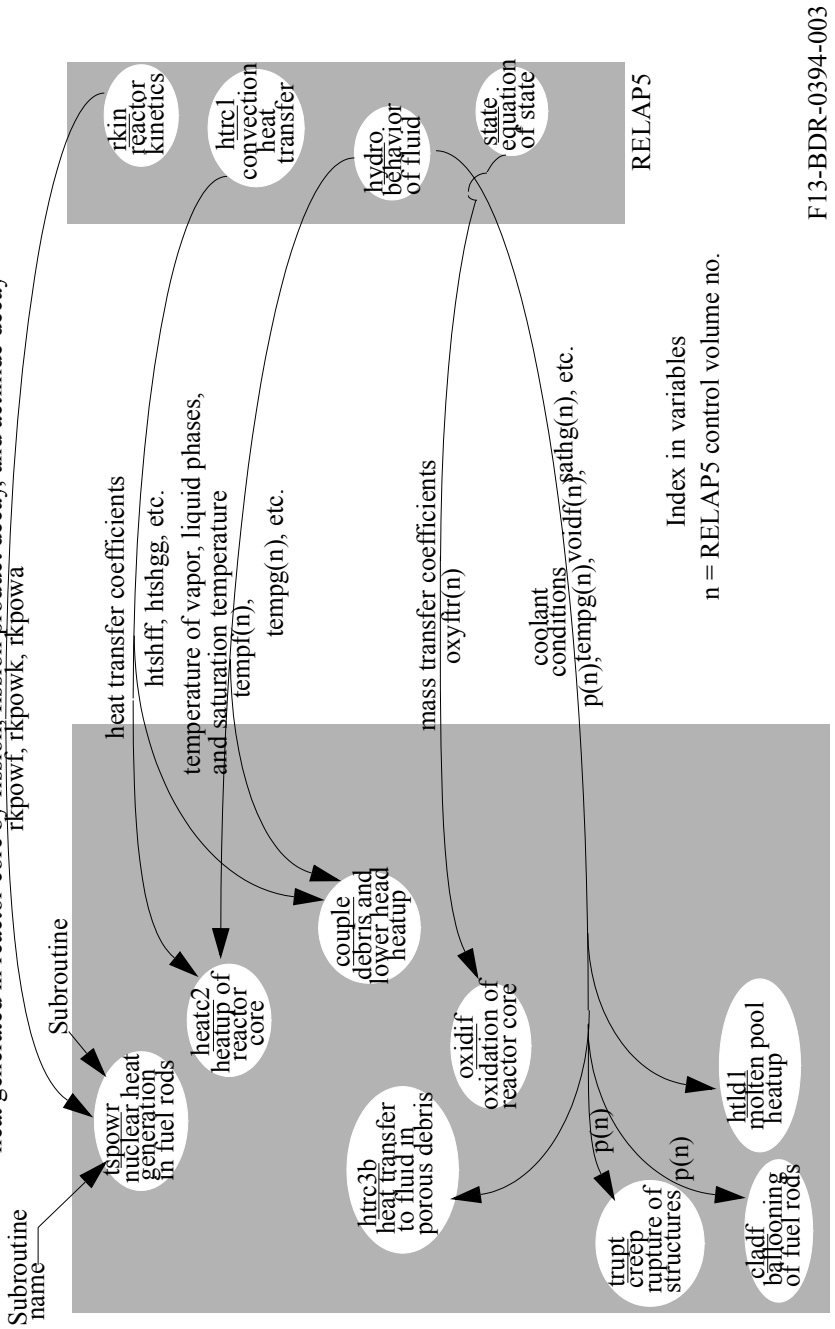

Figure 3-2. Flow of Information from RELAP5 to SCDAP.

temperatures of SCDAP heat structures are calculated in its heat conduction model, which is programmed in a subroutine named HEATC2. The surface temperatures are calculated for each axial node of each component of the reactor core. The surface temperatures of the COUPLE heat structures are calculated in the subroutine named GELB. The surface temperatures for each COUPLE node that interfaces with fluid are passed on to RELAP5. The surface temperatures are used by the RELAP5 subroutine named HTRC1 to calculate for each heat structure the convective heat transfer between the surfaces of the heat structure and the liquid and vapor phases of the fluid interfacing with the heat structure. The volumetric generation rate of vapor at the surface of each heat structure and the heat transfer coefficient to the vapor and liquid phases of water are also calculated by subroutine HTRC1. The variables calculated by HTRC1 for heat structures are calculated by HTRC3B for porous debris.

The heat transferred by radiation from the surfaces of SCDAP and COUPLE heat structures to the fluid interfacing with these heat structures is calculated by SCDAP and passed on to RELAP5 in order to be included in calculation of the internal energy and temperature of the fluid interfacing with these structures. The radiation heat transfer from the SCDAP heat structures to the interfacing fluid is calculated in the subroutine named RADCC2. The radiation heat transfer is stored in the variable named qscd(j,k), where the index “j” is the axial node number, and the index “k” is the component number. The variable qscd also stores the convective heat transfer. This variable is also updated in subroutine SCDAD4 to account for heat transferred to the fluid by gamma radiation from SCDAP heat structures.

The degradation of the reactor core has a significant impact on the flow of coolant through the region of the reactor core and the region of the lower head. The ballooning of fuel rods may result in the flow area at the location of the ballooned fuel rods being reduced by more than 50%. The metallic meltdown of fuel rods may result in a 100% blockage to flow in the axial direction at the locations where the slumping material solidifies. The fragmentation of embrittled fuel rods that quench may result in a factor of one hundred increase in the flow resistance at the locations with fragmented fuel rods. The melting of ceramic material may result in a 100% blockage to flow in the axial and lateral directions. The remelting and slumping of blockages results in the resumption of flow at locations where it previously did not occur due to the blockages. The flow areas, volumes, volume lengths, and hydraulic diameters of RELAP5 control volumes in the core region and lower head are adjusted by SCDAP to account for the changes in geometry caused by fuel rod ballooning, fuel rod and control rod meltdown, fragmentation of embrittled fuel rods, and slumping of core material to the lower head of the reactor vessel. The RELAP5 variables updated by SCDAP are: (1) ajun(l), which are the junction areas, (2) avol(n), which are the control volume areas, (3) v(n), which are the volumes of control volumes, (4) dl(n), which are the lengths of control volumes, and (5) diamv(n), which are the hydraulic diameters of control volumes. The index “l” refers to the junction index and the index “n” refers to the volume index. The junction areas in both the axial and lateral directions are adjusted. These RELAP5 variables are adjusted in subroutine SCDAD4 to account for degradation of the reactor core and are adjusted in subroutine SCDAD7 to account for the filling of the lower head with slumping material from the core region.

The RELAP5 variable that stores for each control volume the source term of noncondensable gases is adjusted by SCDAP to account for the production of hydrogen by the oxidation of the reactor core and to account for the release of noncondensable fission gases from fuel rods. The RELAP5 variable updated by

SCDAP is gaman(ng,n), which is the volumetric source term for the ng-th type of noncondensable gas for the control volume with the index "n". This variable is updated in subroutines OXIDIF (fuel rod oxidation), OXDCON (PWR control rod oxidation), BWHTCN (BWR control blade oxidation), BLADRV (BWR control blade oxidation), and FGRELG (fission gas release and fill gas from fuel rods). The source terms for several types of gases are calculated by SCDAP and passed on to RELAP5. The most important noncondensable gas is hydrogen. Other noncondensable gases are Xe, Kr, and He, which is a fill gas for fuel rods. These source terms are updated by SCDAP at each time step for each axial node of each core component.

SCDAP updates three RELAP5 variables to account for the heat transferred from porous debris to fluid, the heat transferred from the outer surface of a molten pool to the fluid in contact with the molten pool, and the heat transferred from slumping debris that breaks up. The variables updated by SCDAP are q(n), qwg(n), and gammaw(n), where q(n) is the total heat transferred to the fluid in the RELAP5 control volume with the index "n", qwg(n) is the heat transferred to the vapor in this control volume, and gammaw(n) is the volumetric vapor generation rate in this control volume. These variables are updated in subroutine HTRC3B to account for the heat transferred from porous debris to the fluid in the porous debris. These variables are updated in subroutine SCDPRH to account for the heat transferred from the surfaces of intact SCDAP heat structures and the heat transferred from the surfaces of a molten pool to the fluid in contact with the molten pool. These variables are also updated in subroutine SCDPRH to account for the heat transferred from slumping material to the fluid in contact with the slumping material.

RELAP5 calculates several variables that are transferred to SCDAP to calculate the behavior of the reactor core. The first variable transferred to SCDAP is the power of the reactor core. This variable has three parts; namely rkpowf(i), which is the power due to fission, rkpowk(i), which is the power due to fission product decay, and rkpowa(i), which is the power due to actinide decay. The index "i" is defined by the file manager for RELAP5. The SCDAP subroutine named TSPOWR uses the RELAP5 calculated power to calculate the power in each SCDAP heat structure. The next nine variables transferred to SCDAP from RELAP5 define fluid conditions for calculating convective heat transfer. The first variable in this set of nine variables is the heat transfer coefficient from wall to liquid using the liquid temperature. The second variable in this set is the heat transfer coefficient from wall to liquid using saturation temperature corresponding to total pressure. The third variable in this set is the heat transfer coefficient from wall to vapor using the vapor temperature. The fourth variable in this set is the heat transfer coefficient from wall to vapor using saturation pressure corresponding to total pressure. The fifth variable in this set is the heat transfer coefficient from wall to vapor using saturation temperature corresponding to partial pressure of the steam. These five variables are calculated by the RELAP5 subroutine HTRC1. The sixth variable in this set is the temperature of the liquid phase. The seventh variable is the temperature of the vapor phase. The eighth variable is the saturation temperature corresponding to total pressure. The ninth variable in this set is the saturation temperature corresponding to the partial pressure of steam. These nine variables are calculated by the RELAP5 subroutine HYDRO. These nine variable are named htshff(n), htshft(n), htshgg(n), htshft(n), htshgp(n), tempf(l), tempg(l), tsatt(l), and satt(l), where n is the index identifying the number of the SCDAP component and its axial node, and l is the volume index of the RELAP5 control volume that interfaces with the location defined by index n. These nine variables are transferred for each axial node of each SCDAP heat structure and for the nodes at the surface of each COUPLE heat structure.

The heat transfer coefficients and fluid temperatures are used to define a boundary condition in SCDAP and COUPLE heat conduction models. Another variable transferred to SCDAP is the rate of mass transfer of steam through the boundary layer at the surface of SCDAP heat structures that are oxidizing. This variable is named oxyftr(n), where the index "n" is the number of the RELAP5 control volume. This variable is transferred for each axial node of each SCDAP heat structure. The variable is used to define a boundary condition in the oxidation models. Another variable transferred to SCDAP is the fluid pressure p(n). This variable is used by SCDAP to define a boundary condition in the fuel rod ballooning model, calculate mass transfer coefficients for the oxidation model, and to define a boundary condition in the model for creep rupture of the lower head and other structures. Finally, other thermodynamic and transport properties of fluid are transferred to SCDAP to calculate heat transfer from porous debris to fluid, heat transfer from a molten pool to fluid, and heat transfer from slumping material to fluid.

3.3 SCDAP Control of RELAP5 Processes

A number of SCDAP phenomena have an impact on RELAP5 processes. These phenomena are described in the following section.

3.3.1 Changes in Hydrodynamic Characteristics Due to Reactor Core Damage

Two potential phenomena can cause a change in the hydrodynamic flow area used by RELAP5. These phenomena are core component ballooning and corium meltdown/relocation. Core component ballooning can occur in a fuel rod during a high temperature accident. The meltdown and relocation of corium occurs when material from a control rod or fuel rod melts and relocates to a region lower in the core creating a plane blockage.

The reduction of hydrodynamic volume flow area caused by cladding ballooning is accomplished by reducing the value of the variable in which flow area is stored. Because this is a permanent change, unlike flow blockage where the blockage can remelt and move lower, no additional information regarding the flow area prior to flow blockage is required. This reduction in flow area impacts flow in both the axial and crossflow directions.

The slumping of melted fuel rod cladding causes a reduction in flow area at the locations at which the slumped material freezes. If a sufficient amount of material freezes at a location, a plane blockage results. A schematic of this type of blockage to fluid flow is shown in Figure 3-3. The figure shows the part of a reactor core represented by two flow channels and three axial nodes. The left part of the figure shows the state of the core prior to a blockage, and the right part shows the state after a blockage. The blockage occurs in flow channel 1 of axial node 2 and is caused by the slumping of a liquefied mixture of cladding and dissolved fuel. The overlapping of slumped, frozen material causes a plane blockage to occur at an axial node before that node is completely filled with slumped material. The blockage blocks flow in the axial direction at axial node 2 of flow channel 1, but only partially blocks flow in the lateral direction at axial node 2 of flow channel 1. Before the plane blockage is imposed, the loss coefficients increase and the flow area decreases at axial node 2 as a function of the amount of slumped material in axial node 2. This smoothing of the transition in configuration is described in detail in Volume II.

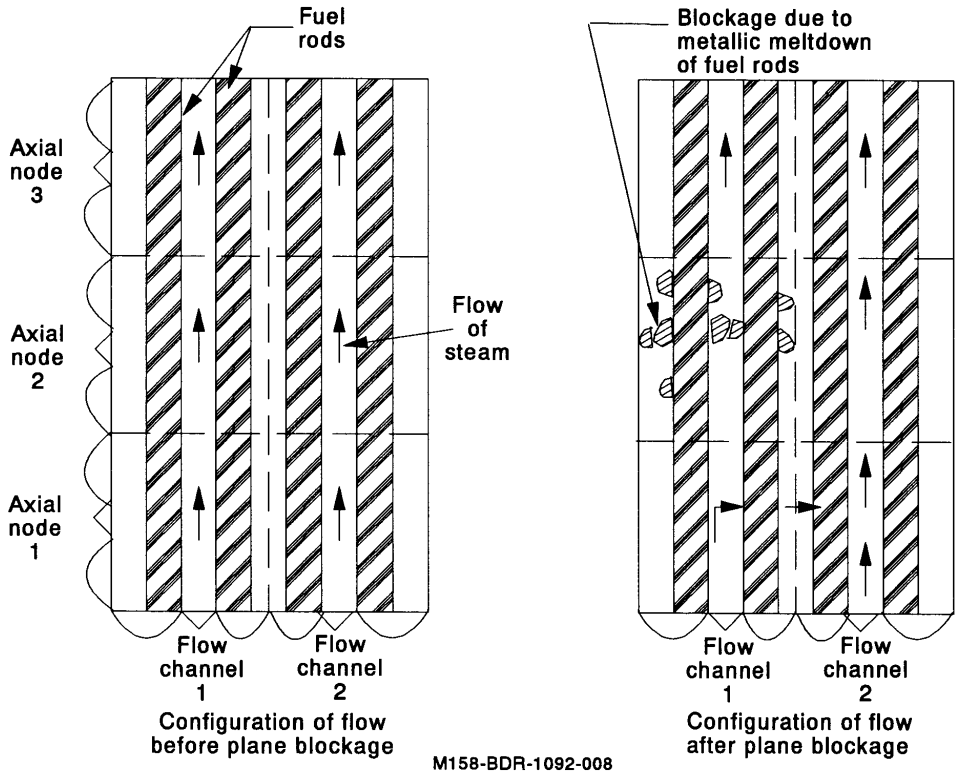

Figure 3-3. Schematic of plane blockage caused by metallic meltdown.

Blockages are removed at locations where the blockage material slumps again. Both plane blockages (Figure 3-3) and bulk blockages (Figure 3-4) may be removed. In the case of removing plane blockage, the junction flow area at the location of the blockage changes from zero to the flow area of the junction prior to the blockage. When a bulk blockage is removed, the junction flow areas for the volume are restored to their pre-blockage values and the RELAP5 control volume modeling the fluid at that location is activated.

Bulk blockages are caused by the melting of the UO_2 and ZrO_2 in fuel rods. These constituents are the materials in a fuel rod with the highest melting temperature. As a result, their melting causes a complete loss of structural framework and the conglomeration of a relatively large amount of material with no porosity. The slumping of the conglomerated material is stopped after it has moved into a cooler location and its bottom and side boundaries freeze. A bulk blockage blocks flow in both the axial and lateral directions at a node. The transition in configuration is smoothed as for a plane blockage. If a bulk blockage slumps away from a location, a calculation is again made of the amount and composition of the fluid at that location and of the thermodynamic state of the fluid.

The volume available for fluid flow is increased at locations where melting of porous material results in a compaction and settling of the molten material. The change in configuration represented by this process is shown in Figure 3-5. The left part of the figure shows porous debris that is beginning to melt at

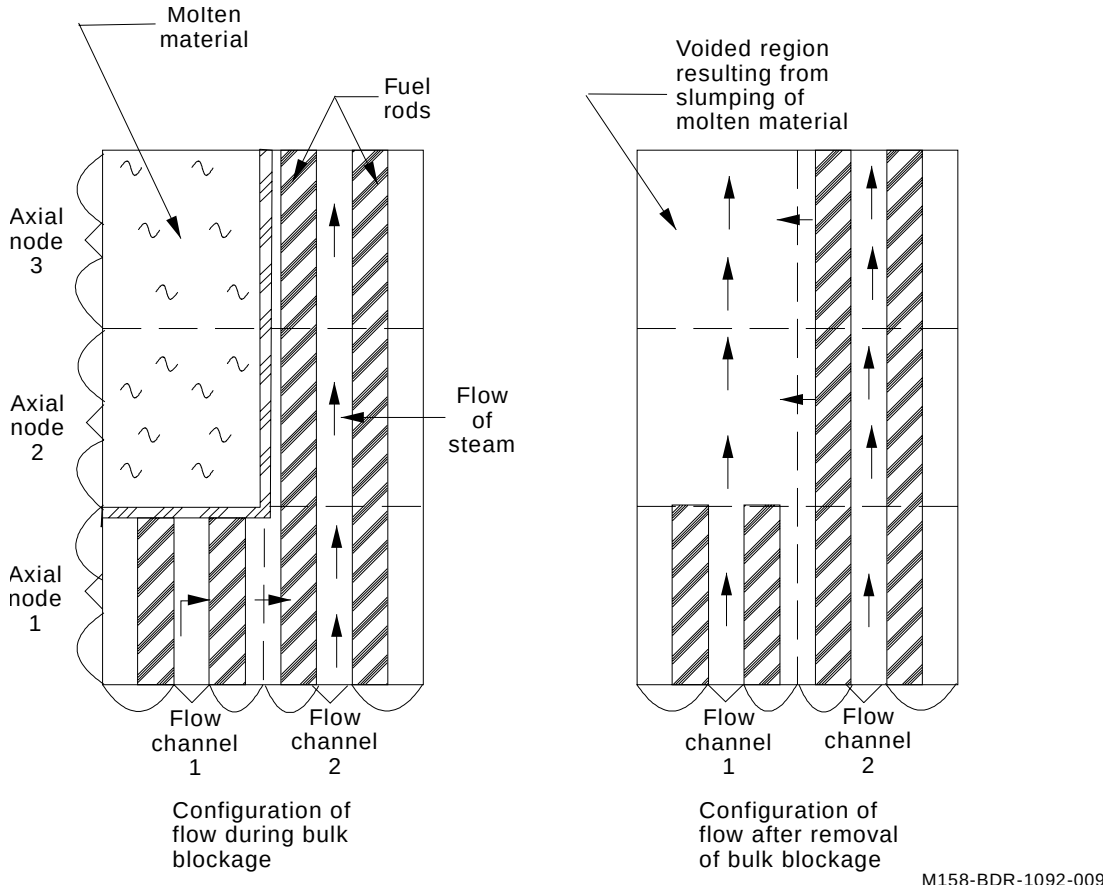

Figure 3-4. Schematic of blockage removal.

axial nodes 2 and 3 of flow channel 1. Steam is flowing through both of these axial nodes. The right part of the figure shows the pattern of steam flow after the porous debris at the two axial nodes has completely melted. The melting porous debris at axial node 3 slumps into the open porosity at axial node 2. To represent this change in configuration, the blockage model assigns a flow area of zero to all of the RELAP5 junctions connected with axial node 2 and deactivates the RELAP5 control volume representing the fluid at axial node 2. For axial node 3, the model assigns the RELAP5 control volume for this location a volume that represents the increase in space resulting from the material at this location slumping to axial node 2.

A significant increase in flow losses and a significant change in the configuration of flow occurs at a location in the reactor core that disintegrate from the configuration of an array of vertical rods to the configuration of porous debris. A schematic of this change is shown in Figure 3-6. After this kind of disintegration has occurred, the path for fluid flow becomes much more tortuous. As a result, principles for flow loss applicable for flow through a vertical array of rods cannot be applied. Instead, Darcy's Law for porous debris is applied. For locations with porous debris, the size of the debris particles and the porosity of the debris are calculated by SCDAP (subroutine rubtrn) and conveyed to RELAP5 (subroutine hloss). The permeability and passability of the porous debris are then calculated using these debris characteristics

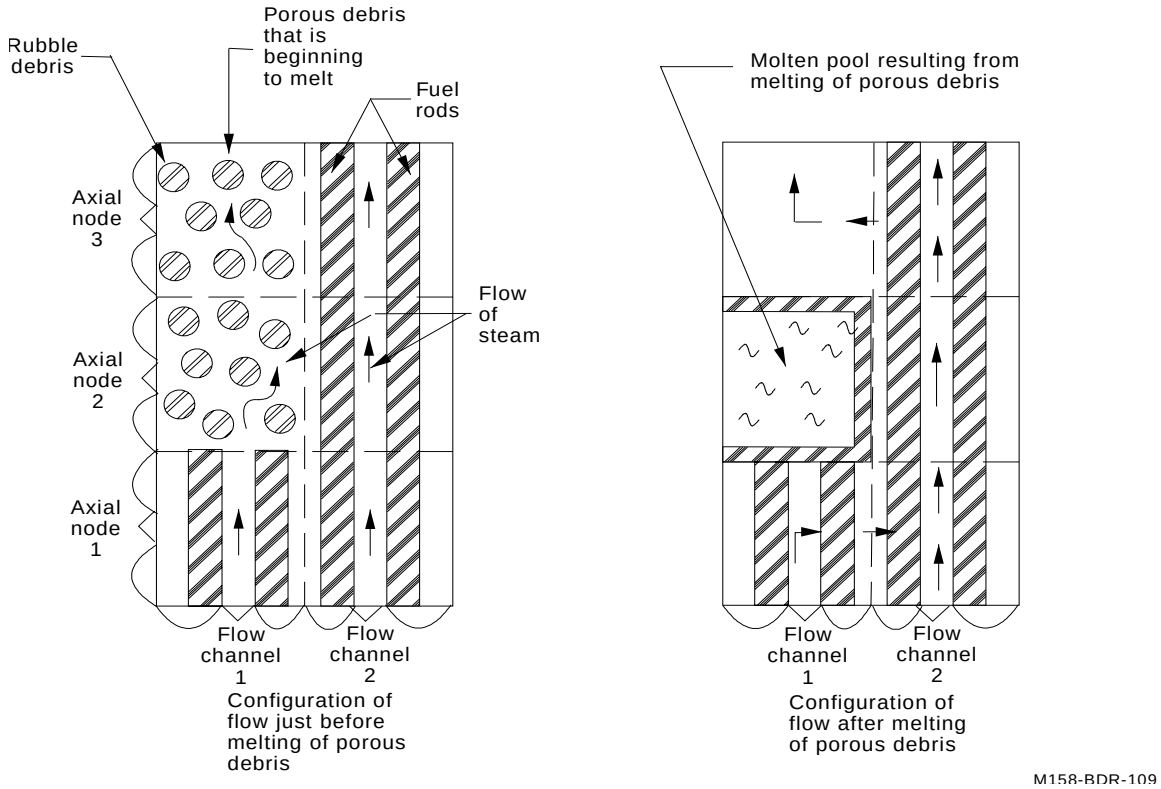

Figure 3-5. Affect of melting of porous debris on flow configuration.

and the volume fractions of liquid and vapor phase of coolant at the location. The flow losses are then calculated as a function of the permeability and passability.

3.3.2 Noncondensable Transport

The SCDAP portion of the code has the capability of releasing noncondensable gases (hydrogen and fission products). The RELAP5 portion of the code must track the migration of these gases through the primary coolant system, and evaluate their impact on the thermal response of the remainder of the system. As a minimum therefore the code user must specify at least hydrogen as a noncondensable on input, in order to force RELAP5 to allocate storage for noncondensable. As documented in Volume 4 of Reference 10, the effects of a noncondensable within RELAP5 are represented by multipliers that modify the volumetric heat transfer coefficients. In sufficient quantities the presence of a noncondensable may also impact flow regime determination.

3.4 RELAP5 Calculations Conveyed to SCDAP

RELAP5 calculations of convective cooling and coolant conditions are conveyed to SCDAP for its calculation of the heatup and damage of the reactor core and lower head. The SCDAP heat conduction model for intact reactor core structures uses as boundary conditions the convective heat transfer coefficients for the structures calculated by RELAP5. The SCDAP model for radiation heat transfer uses the RELAP5 calculations of coolant pressure, and emissivities and absorptivities of the vapor phase. The

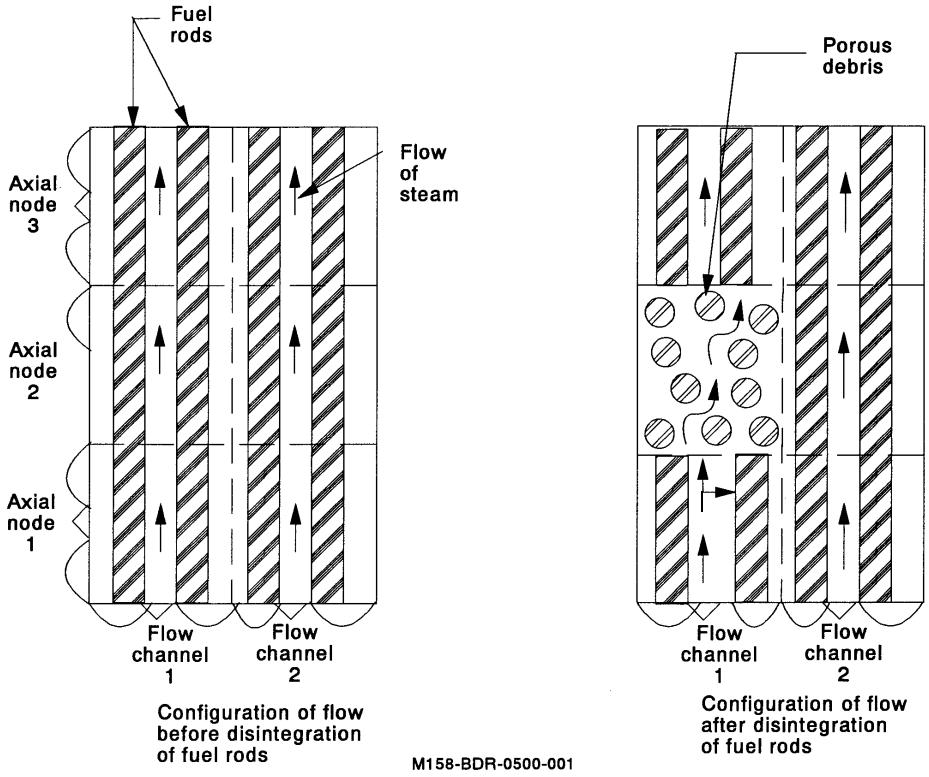

Figure 3-6. Schematic of change in configuration of flow caused by disintegration of fuel rods into porous debris.

SCDAP model for the convective and radiative cooling of porous debris uses the coolant conditions and coolant properties calculated for the porous debris locations by RELAP5. These coolant conditions and properties include; (1) volume fractions of liquid and vapor, (2) fluid pressure, (3) velocities of liquid and vapor phases, (4) densities of liquid and vapor phases, and (5) viscosities of liquid and vapor phases. The SCDAP model for oxidation uses the RELAP5 calculations of coolant pressure and density, noncondensible quality, mass diffusivity of H_2O in a gas mixture, and convective heat transfer coefficient. The SCDAP model for cladding ballooning and lower head creep damage uses the coolant pressure calculated by RELAP5. The SCDAP heat conduction model for intact reactor core structures and its heatup models for porous debris and molten pools use the RELAP5 calculations of the heat generation due to fission and decay heat.

4. SCDAP/RELAP5-3D© EXTENSIONS TO RELAP5 SYSTEM MODELS

This section describes extensions to existing RELAP5 models to transform them into SCDAP/ RELAP5-3D© specific models.

4.1 Reactor Kinetics Model

The RELAP5 reactor kinetics model is extended to SCDAP/RELAP5-3D© by allowing SCDAP component temperatures to impact the temperature feedback, and allowing the resultant power in SCDAP components.

One of two models can be selected for reactivity feedback. One model assumes nonlinear feedback effects from moderator density and fuel temperature changes and linear feedback from moderator temperature changes. It is called the separable model because each effect is assumed to be independent of the other effects. Boron feedback is not provided, but a user-defined boron feedback can be implemented with the control system. The separable model can be used if boron changes are quite small and the reactor is near critical about only one state point.

The separable model defines reactivity as

$$r(t) = r_0 - r_B + \sum_{i}^{s} r_{si}(t) + \sum_{i}^{c} V_{ci} + \sum_{i}^{p} \{W_{\rho i} R_\rho[\rho_i(t)] + a_{Wi} T_{Wi}(t)\} + \sum_{}^{n_F} \{W_{Fi} R_F[T_{Fi}(t)] + a_{Fi} T_{Fi}(t)\} \quad . \tag{4-1}$$

The quantity r_0 is an input quantity that represents the reactivity corresponding to assumed steady-state reactor power at t = 0. The quantity r_B is the bias reactivity, which is calculated during input processing such that $r(0) = r_0$. The quantities r_{si} are obtained from input tables defining n_s reactivity curves as a function of time. The quantities V_{ci} and n_c are control variables that can be user-defined as reactivity contributions. R_ρ is a table defining reactivity as a function of the current density of water ρ_i (t) in the hydrodynamic volume i; $W_{\rho i}$ is the density weighting factor for volume i; T_{wi} is the spatial density averaged temperature of volume i; a_{Wi} is the temperature coefficient (not including density changes) for volume i; and n_p is the number of hydrodynamic volumes in the reactor core. The value R_F is a table defining reactivity as a function of the average fuel temperature T_{Fi} in a SCDAP fuel rod component; W_{Fi} and a_{Fi} are the fuel temperature weighting factor and the fuel temperature coefficient, respectively; and n_F is the number of SCDAP fuel rod components in the reactor core.

As previously mentioned this model is applicable if the reactor is near critical about only one state point. However, a postulated BWR anticipated transient without scram (ATWS) accident is an example

where the reactor could be nearly critical for two different state points. One point is at normal power operating conditions-high moderator and fuel temperatures, highly voided, and no boron. During accident recovery, the reactor might approach a critical condition with relatively cold moderator and fuel temperatures, with no voids, but with some boron concentration. The reactivity could be nearly critical for both states, but the contributions from the different feedback effects are vastly different. The assumptions of no interactions among the different feedback mechanisms, especially boron, cannot be justified. In this case the tabular model would be more applicable.

The tabular model defines reactivity as

$$r(t) = r_0 - r_B + \sum_{i}^{n_s} \sigma_{si} + \sum_{i}^{n_c} V_{ci} + R[\rho(t), T_w(t), T_F(t), B(t)] \quad (4\text{-}2)$$

$$\rho(t) = \sum_{i}^{n_\rho} W_{\rho i} \rho_i(t) \quad (4\text{-}3)$$

$$T_w(t) = \sum_{i}^{n_\rho} W_{\rho i} T_{Wi}(t) \quad (4\text{-}4)$$

$$B(t) = \sum_{i}^{n_F} W_{\rho i} B_i(t) \quad (4\text{-}5)$$

$$T_F(t) = \sum_{i} W_{Fi} T_{Fi}(t) \quad (4\text{-}6)$$

where B is boron density. The average quantities are obtained with the use of one weighting factor for each hydrodynamic volume and each SCDAP component contributing to reactivity feedback. The reactivity function R is defined by a table input by the user. In the TABLE4 option the table is four-dimensional; the TABLE3 option assumes no boron dependence and the table is then three-dimensional. Although the tabular model overcomes the objections of the separable model, because all feedback mechanisms can be nonlinear and interactions among the mechanisms are included, the penalty for the expanded modeling capability greatly increases the input data requirements.

4.2 Nuclear Heat Model

For each SCDAP component, the user may select either the SCDAP model or one of the RELAP5 power options to calculate nuclear heat. The following discussion refers to the SCDAP nuclear heat model.

The energy release from fissioning of uranium is manifested in a variety of forms. Table 4-1 lists recoverable forms of energy from thermal fissioning of ^{235}U that are considered in the nuclear heat model and average energy per fission associated with each form. The Evaluated Nuclear Data File[11] is the source of information for the first four forms of energy in Table 4-1, and the ANSI/ANS-5.1-1979 decay heat standard[12] is the source for the fifth form of energy listed. The first three forms of energy in Table 4-1 are released at the time of the fission event and are collectively known as prompt nuclear heat. They account for about 92% of the recoverable fission energy. The remaining two forms of energy in Table 4-1 appear at some time after the fission event and are classified as delayed nuclear heat. This energy is due to the

radioactive decay of a large number of different fission products and actinides (^{239}U and ^{239}Np in this model). Since radioactive decay is a random event that is characterized by an exponential probability distribution, the delayed nuclear heat is exponentially distributed in time following a fission event.

Table 4-1. Energy release from fission of ^{235}U.[a]

	Form	Energy (MeV/fission)
I	Kinetic energy of fission products (prompt)	169.58
II	Fission γ-rays (prompt)	6.96
III	Kinetic energy of fission neutrons (prompt)	4.79
IV	Fission product decay (delayed)	
	β	6.43
	γ	6.26
	Neutrons (kinetic energy)	0.0071
V	Neutron capture activation (delayed)	
	Fission Products	~.5
	Actinides	~.8

a. Total from forms I, II, and III = 181.33 MeV/fission (prompt). Total from forms IV and V = 14 MeV/fission (delayed, recoverable). Energy from neutrons is nonrecoverable.

Prompt nuclear heat is determined by thermal neutron flux levels. Because neutron flux distributions are not determined by the code, the nuclear heat model requires that any prompt nuclear heat generated during the problem time be input to the model. Delayed nuclear heat is determined by the prior operating history. This is provided as input to the model by specifying the rod average total (prompt plus delayed) volumetric power for each time span prior to the start of the problem. For nonfuel rod components (i.e., control rods or structural material), the user must supply total power for the problem time period. For fuel rods, the user has the option of supplying total power for the problem time period and thus overriding the decay heat calculations. Additional required input information are fuel enrichment, fuel density, rate of ^{239}U production (ratio of fission producing neutron absorption in ^{238}U), and axial and radial multiplication factors that convert the rod average prompt and delayed power to local power.

This model determines the delayed nuclear heat based on the ANSI/ANS-5.1-1979 decay heat standard, and the required nuclear data are based on the Evaluated Nuclear Data File. Neither the delayed nuclear heat from neutron capture in structural material nor the energy added by fissioning due to delayed neutrons subsequent to reactor shutdown are considered. However, the user can force consideration of these effects by specifying their contributions as an additional prompt nuclear heat source. The delayed nuclear heat model is based on the following major assumptions:

- The fuel rods are of commercial LWR design.
- The bundle component remains intact and at initial density and volume. Movement of

material due to fuel rod disruption is accounted for outside of this model by the SCDAP subroutines FSTATE and CFDAMG, depending on the state of the bundle. These routines maintain an inventory system that will trace a piece of material back to its original location in order to determine the current decay power. (The prompt power is not history dependent; therefore, only the current location peaking factor and average power are required to determine current prompt power.)

- ^{235}U is the only fissile material.
- ^{238}U is the only fertile material. (This implies that the only actinide decay chain that will be considered is the ^{239}U to ^{239}Pu decay chain due to ^{238}U neutron absorption.)
- Gamma energy is completely recoverable with a flat radial distribution and the user-supplied axial distribution of energy deposition.

Most of the computations required for this model, including those for establishing the ANSI/ANS-5.1-1979 based delayed nuclear heat, are performed during initialization. These computations establish power history tables and axial and radial distribution tables. The volumetric power for any time and location within the analysis is determined by interpolating these tables.

The total nuclear heat source due to fissioning is represented as

$$Q(z,r,t) = Q_p(t)\, Z_p(z)\, R_p(r) + Q_d(t)\, Z_d(z)\, R_d(r) \qquad (4\text{-}7)$$

where

$Q(z,r,t)$	=	total nuclear heat at axial position z and radial position r of the component at time t (W/m^3)
$Q_p(t)$	=	component average prompt nuclear heat at time t (W/m^3)
$Z_p(z)$	=	axial peaking factor for prompt heat at position z
$R_p(r)$	=	radial peaking factor for prompt heat at position r
$Q_d(t)$	=	component average delayed nuclear heat at time t (W/m^3)
$Z_d(z)$	=	axial peaking factor for delayed heat at position z
$R_d(r)$	=	radial peaking factor for delayed heat at position r.

For a fuel rod component, the user must supply the three factors of the first term of Equation (4-7) (Q_p, Z_p, R_p) as well as the factor Z_d from the second term of the equation. The user may either supply Q_d or allow the model to calculate the term. For components other than fuel rods, the user must supply Q_d.[a]

a. The user may also supply nonnuclear heat (i.e., electrical) through the terms in Equation (4-7). The delayed nuclear heat model options are not recommended for use in such situations.

The factor R_d is always determined by the model, although for components other than fuel rods, R_d is set equal to R_p.

The factor Z_p accounts for the axial distribution of prompt nuclear heat due to the neutron flux distribution. Since Z_p may shift during the transient, the user can supply separate axial peaking factor arrays for different time periods. Under equilibrium conditions, the distribution of the delayed nuclear heat is the same as the prompt distribution. However, when the prompt distribution changes, the delayed distribution exponentially approaches the new shape with time. Therefore, the delayed distribution may be different from the prompt distribution; the user can input a separate axial nuclear heat distribution for delayed nuclear heat. Since the conditions that cause the radial distribution of prompt heat are not expected to change significantly with time, the user is only allowed to input one prompt radial distribution shape. The model always provides the radial distribution of delayed heat based on the user-supplied prompt radial distribution and the component type (fuel rod or nonfuel rod).

Expanding the second term of Equation (4-7) (delayed nuclear heat) into its components yields

$$Q_d(t)Z_d(z)R_d(r) = (DF \;\; G \;\; Q_{fd} + Q_{Ad})Z_d(z)R_d(r) \tag{4-8}$$

where

Q_{fd} = rod average fission product decay power (W/m^3)

Q_{Ad} = rod average actinide (^{239}U, ^{239}Np) decay power (W/m^3)

DF = reduction factor in decay heat due to loss of volatile fission products[15]

G = neutron capture correction to fission product decay.

A comparison of the magnitude of these components is given in 4-1. The reduction in decay power due to loss of volatile fission products is discussed in Reference 15. The remaining components are discussed in the following sections of this report.

4.2.1 Fission Product Decay Power

Reference 12 provides an empirical correlation for the decay power per fission t seconds after a fission pulse from the thermal fission of ^{235}U. The form of the correlation is given by

$$f(t) = \sum_{i=1}^{23} \alpha_i e^{(-\lambda_i t)} \tag{4-9}$$

where

F(t) = decay power per fission t seconds after a ^{235}U thermal fission pulse (MeV/fission • s)

α_i = initial decay value of empirical (nonphysical) Group i (MeV/fission • s)

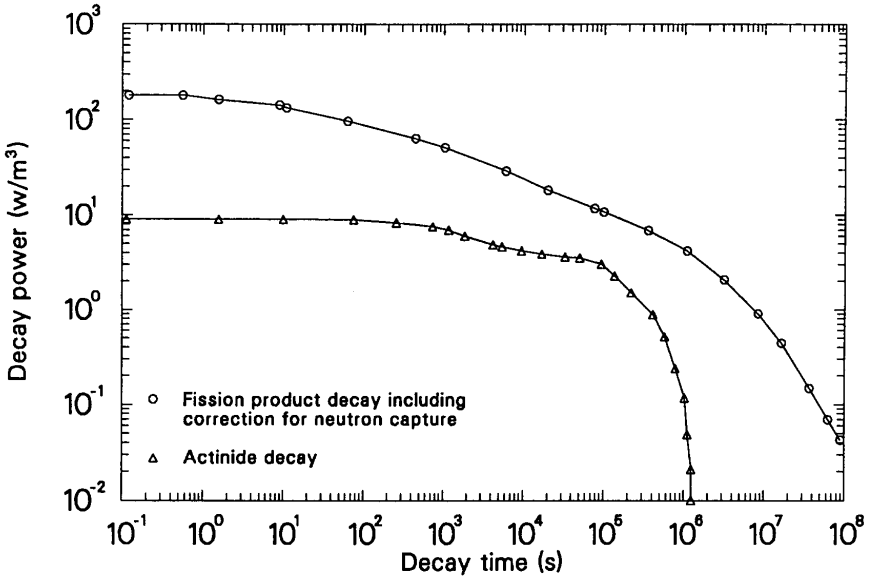

Figure 4-1. Comparison of decay heat components.

λ_i = time-constant of empirical Group i (s^{-1}).

The constants α_i and λ_i are given in Table 4-2.

Table 4-2. Decay power correlation constants .

i	α_i	λ_i	i	α_i	λ_i
1	6.5057E-1	2.2138 E+1	13	2.5232 E-6	1.0010 E-5
2	5.1264 E-1	5.1587 E-1	14	4.9948 E-7	2.5438 E-6
3	2.4384 E-1	1.9594 E-1	15	1.8531 E-7	6.6361 E-7
4	1.3850 E-1	1.0312 E-1	16	2.6608 E-8	1.2290 E-8
5	5.5440 E-2	3.3656 E-2	17	2.2398 E-9	2.7213 E-8
6	2.2225 E-2	1.1681 E-2	18	8.1641 E-12	4.3714 E-9
7	3.3088 E-3	3.5870 E-3	19	8.7797 E-11	7.5780 E-10
8	9.3015 E-4	1.3930 E-3	20	2.5131 E-14	2.4786 E-10
9	8.0943 E-4	6.2630 E-4	21	3.2176 E-16	2.2384 E-13
10	1.9567 E-4	1.8906 E-4	22	4.5028 E-17	2.4600 E-14
11	3.2535 E-5	5.4988 E-5	23	7.4791 E-17	1.5699 E-14
12	7.5595 E-6	2.0958 E-5			

The decay power after an operating period is the product of Equation (4-9) and the fission rate integrated over the operating period T:

$$Q_{fd}(T) = \int_0^T f(T - T')FR(T')dT' \tag{4-10}$$

where FR(T') is the fission rate of ^{235}U at time T' (fission/m^3 • s) (W/MeV).

The term FR(T') is determined by

$$FR(T') = \frac{Q_p(T')}{E_f} = \frac{Q(T') - Q_d(T')}{E_f} \tag{4-11}$$

where

$Q_p(T')$ = prompt nuclear heat at time T' (W/m^3)

$Q(T')$ = total nuclear heat at time T' (W/m^3)

$Q_d(T')$ = decay nuclear heat at time T' (W/m^3)

E_f = prompt energy per fission of ^{235}U = 181.33 (MeV/fission).[15]

Substituting Equation (4-9) into Equation (4-10) and breaking the integration into a sum of operating periods for which FR is constant yields

$$Q_{fd}(T) = \sum_{n=1}^{N} FR_n \left[\sum_{i=1}^{23} \alpha_i \int_{T_{n-1}}^{T_n} e^{-\lambda_i (T - T')} dT' \right] \tag{4-12}$$

where N is the number of periods used to represent the time span 0 to T.

Integrating Equation (4-12) yields

$$Q_{fd}(T) = \sum_{n=1}^{N} FR_n \left[\sum_{i=1}^{23} \frac{\alpha_i}{\lambda_i} e^{(-\lambda_i T)} [1 - e^{(-\lambda_i \Delta t_n)}] \right] \tag{4-13}$$

where

$$\Delta t_n = T_n - T_{n-1} \text{ (s)}. \tag{4-14}$$

By expanding Equation (4-13), rearranging terms, and separating the decay heat into 23 components, the following may be obtained:

$$Q^i_{fd,\,n} = Q^i_{fd,\,n-1} e^{(-\lambda_i \Delta t_n)} + FR_n \frac{\alpha_i}{\lambda_i} [1 - e^{(-\lambda_i \Delta t_n)}] \tag{4-15}$$

where

$Q^{i}_{fd,\,n}$ = i-th component of decay heat after time step n (W/m^3)

$Q^{i}_{fd,\,n-1}$ = i-th component of decay heat after time step n-1 (W/m^3).

The fission product decay heat is now in a form that can be updated each time step given the current fission rate and time step size.

4.2.2 Neutron Capture Correction to Fission Product Decay

Reference 12 provides two empirical methods for determining the neutron capture correction to fission product delay [the factor G from Equation (4-7)]. One method is a table of values that is valid for shutdown times up to 10^9 seconds and operating times up to 4 years for standard LWR operating conditions. The other method is a correlation that is valid for shutdown times up to 10^4 seconds, operating times up to 4 years, and fissions per initial fissile atom up to three. The correlation is used for times less than 10^4 seconds, since it is more accurate within this range; and the tables are used for times greater than 10^4 seconds. The correlation is given by

$$G = 1.0 + (3.24 \times 10^{-6} + 5.23 \times 10^{-10} t)\, T^{0.4} \psi \qquad (4\text{-}16)$$

where

G = neutron capture correction to fission product decay

t = time since shutdown (s)

T = operating time (s)

ψ = fissions per initial fissile atom.

The term ψ is determined from the operating history by

$$\Psi = \frac{1}{\nu}\int_0^T FR\, dT' \qquad (4\text{-}17)$$

where

ν = number of initial fissile atoms per unit volume ($atoms/m^3$)

FR = fission rate per volume ($fissions/m^3 \cdot s$).

The term ν is determined from the fuel pellet initial conditions by

$$\nu = \frac{w \rho N_o}{A t} \qquad (4\text{-}18)$$

where

w	=	weight fraction of $^{235}UO_2$
ρ	=	fuel pellet density (kg/m^3)
N_o	=	Avogadro's number, 6.023×10^{26} (molecule/kg•mole)
At	=	atomic weight of $^{235}UO_2$.

The table of values that are interpolated when conditions are outside the range of Equation (4-16) is shown in Table 4-3.

Table 4-3. G factors for times greater than 10,000 seconds.

Time after shutdown (s)	G (unitless)
1.0 E4	1.064
1.5 E4	1.074
2.0 E4	1.081
4.0 E4	1.098
6.0 E4	1.111
8.0 E4	1.119
1.0 E5	1.124
1.5 E5	1.130
2.0 E5	1.131
4.0 E5	1.126
6.0 E5	1.124
8.0 E5	1.123
1.0 E6	1.124
1.5 E6	1.125
2.0 E6	1.127
4.0 E6	1.134
6.0 E6	1.146
8.0 E6	1.162
1.0E7	1.181
1.5 E7	1.233

Table 4-3. G factors for times greater than 10,000 seconds. (Continued)

Time after shutdown (s)	G (unitless)
2.0 E7	1.284
4.0 E7	1.444
6.0 E7	1.535
8.0 E7	1.586
1.0 E8	1.598
1.5 E8	1.498
2.0 E8	1.343
4.0 E8	1.065
6.0 E8	1.021
8.0 E8	1.012

4.2.3 Actinide Decay Power

The actinide decay term in Equation (4-8) is calculated using the model supplied in Reference 12. This model assumes that the actinide decay power is due only to the decay of ^{239}U and ^{239}Np. The actinide power is given by

$$Q_{Ad} = \frac{Q_f}{E_f}[F_{239_U}(t, T) + F_{239_{Np}}(t, T)] \quad (4\text{-}19)$$

where

Q_f = total maximum[a] fission power during the operating history (W/m^3)

E_f = effective energy release per fission (195.3 MeV/fission)

F = ^{239}U or ^{239}Np decay energy per fission (MeV/fission).

The terms F_{239_U} and $F_{239_{Np}}$ are given by

$$F_{239_U}(t, T) = RE_{239_U}(1 - e^{-\lambda_1 T})e^{-\lambda_1 t} \quad (4\text{-}20)$$

$$F_{239_{Np}}(t, T) = RE_{239_{Np}}\frac{\lambda_1}{\lambda_1 - \lambda_2}(1 - e^{-\lambda_2 T})e^{-\lambda_2 t} - \frac{\lambda_2}{\lambda_1 - \lambda_2}(1 - e^{-\lambda_1 T})e^{-\lambda_1 t} \quad (4\text{-}21)$$

a. Use of total maximum fission power is specified by Reference 15.

where

R	=	^{239}U of production per fission (atoms/fission)
T	=	operating time (s)
t	=	time since shutdown (s)
E_{239_U}	=	average decay energy of ^{239}U (0.474 MeV)
$E_{239_{Np}}$	=	average decay energy of ^{239}Np (0.419 MeV)
λ_1	=	^{239}U decay constant (4.91 E-4 s^{-1})
λ_2	=	^{239}Np decay constant (3.42 E-6 s^{-1}).

4.2.4 Radial Peaking Factor for Delayed Heat

For nonfuel components, the radial distribution of delayed power, $R_d(r)$, is set equal to radial distribution of prompt power, $R_p(r)$. However, for fuel rods, $R_d(r)$ is not identical to $R_p(r)$ because γ-ray energy may be deposited a considerable distance from the location of generation. Since γ-ray energy is about one-half of the delayed power, $R_d(r)$ within a fuel rod is estimated by

$$R_d(r) = 0.5[1 + R_p(r)] \quad . \tag{4-22}$$

The overall radial power distribution is determined by

$$R^j = \frac{(GQ_{fd} + Q_{Ad})Z_d^j(1 + R_p)\sqrt{1 + R_p} + Q_p Z_p^j R_p}{(GQ_{fd} + Q_{Ad})Z_d^j + Q_p Z_p^j} \tag{4-23}$$

where the superscript j denotes axial position and is required due to the possible difference in prompt and delayed axial power distributions.

4.3 Heat Transfer Correlations for External Surface of Lower Head

A set of nucleate boiling heat transfer correlations for a hemispherical surface were implemented to calculate the transfer of heat from a flooded lower vessel head.

4.3.1 Sub-cooled Nucleate Boiling

The set of subcooled nucleate boiling correlations[13,14] were developed for calculation of heat flux (in W/m^2) from a hemispherical surface, and are in the form

$$q = a\Delta T + b\Delta T^2 + c\Delta T^3 \tag{4-24}$$

where

a, b, and c = position-dependent coefficients from Table 4-4 for the orientation shown in 4-2 and

ΔT = the difference between the wall surface temperature and the pool saturation temperature (in K).

Nucleate boiling curves derived from Equation (4-24) are valid from a ΔT of ~4 K to the ΔT associated with the critical heat flux (CHF) for subcooled boiling (in MW/m^2) given by[14]

$$q_{chf} = 0.4(1 + 0.036\Delta T_{sub})(1 + 0.021\theta - (0.007\theta)^2) \quad (4\text{-}25)$$

where

ΔT_{sub} = 10 K (assumed to be a constant in this analysis) and

θ = the surface contact angle in degrees for the orientation shown in 4-2.

Experimental data embodied in Equations (4-24) and (4-25) were then applied to each lower head surface node (numbered 1 through 22 in 4-3) to simulate natural convection, nucleate boiling, and transition boiling regimes as follows.

- Equation (4-24) was used to calculate a position-dependent heat flux for ΔT = 4 K. Linear interpolation between zero and the resulting flux (at ΔT = 4 K) was performed to estimate the flux for any ΔT from 0 to 4 K, which should adequately simulate natural convection to a subcooled pool.
- Equations (4-24) and (4-25) were solved simultaneously to determine the position-dependent ΔT associated with the CHF. (Resulting CHF values are summarized for all surface nodes in Table 4-5.) Equation (4-24) was then applied to determine the subcooled nucleate boiling flux for any ΔT between 4 K and the ΔT at the CHF.
- The transition boiling heat flux was linearly extrapolated from the position-dependent CHF to the minimum flux that could occur assuming a heat transfer coefficient of 375 W/m^2-K. The extrapolation was performed for any ΔT greater than the ΔT at the CHF based on an estimate of the slope associated with transition regime experimental data.

Application of the data is depicted graphically in Figure 4-4 for several positions on the exterior surface of the lower head. Experimental data for the film boiling regime were not implemented. The lack of implementation is not expected to be important because lower head failure can be assumed when heat fluxes from core materials in the lower head are high enough to drive the exterior surface beyond the CHF.

Table 4-4. Subcooled nucleate boiling correlation coefficients as a function of position on the exterior surface of the reactor vessel lower hemispherical head.

Subcooled Boiling Correlation	Data Reported At	Correlation Applied At	Correlation Coefficients		
			a	b	c
1	L/D = 0	$0 < L/D \le 0.1$	0	319	-2.83
2	L/D = 0.20	$0.1 < L/D \le 0.275$	4016	430	-4.13

Table 4-4. Subcooled nucleate boiling correlation coefficients as a function of position on the exterior surface of the reactor vessel lower hemispherical head.

Subcooled Boiling Correlation	Data Reported At	Correlation Applied At	Correlation Coefficients		
			a	b	c
3	L/D = 0.35	$0.275 < L/D \leq 0.425$	0	337	2.61
4	L/D = 0.50	$0.425 < L/D \leq 0.625$	0	891	-9.04
5	L/D = 0.75	$L/D > 0.625$	0	529	0.08

Table 4-5. Subcooled boiling correlations and CHF for exterior surface COUPLE mesh nodes.

Surface Node Number	θ	L/D	Subcooled Boiling Correlation	CHF (MW/m^2)
1	0^o	0	1	0.544
2	9.97^o	0.111	2	0.655
3	19.0^o	0.211	2	0.751
4	24.7^o	0.275	2	0.811
5	30.6^o	0.340	3	0.869
6	35.4^o	0.394	3	0.915
7	40.4^o	0.448	4	0.962
8	45.9^o	0.510	4	1.01
9	51.1^o	0.568	4	1.06
10	55.0^o	0.612	4	1.09
11	59.1^o	0.657	5	1.13
12	63.5^o	0.705	5	1.16
13	68.1^o	0.756	5	1.20
14	72.9^o	0.810	5	1.24
15	77.9^o	0.866	5	1.27
16	83.5^o	0.928	5	1.31
17	$> 90^o$	> 1	5	1.36
18	$> 90^o$	> 1	5	1.36
19	$> 90^o$	> 1	5	1.36
20	$> 90^o$	> 1	5	1.36
21	$> 90^o$	> 1	5	1.36
22	$> 90^o$	> 1	5	1.36

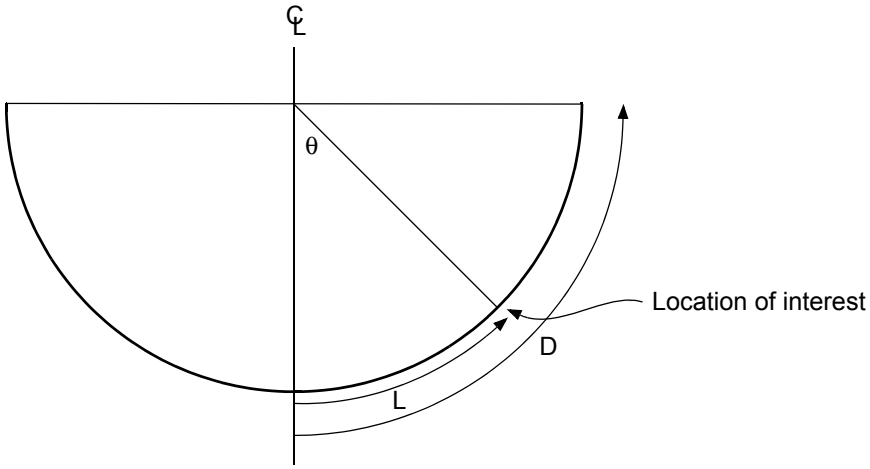

Figure 4-2. Orientation for calculation of boiling heat transfer from a hemispherical surface.

4.4 Special Techniques

4.4.1 Time Step Control

In some earlier versions, SCDAP has not had the capability of repeating the time step used within SCDAP/RELAP5-3D©; whatever time step that was selected for RELAP5 was automatically used in SCDAP. This caused stability problems when, as an example, the SCDAP heat structure is radiating energy into RELAP5 hydrodynamic volume. In this case both SCDAP and RELAP5 need to use a smaller time step determined by the heat flux from SCDAP heat structures into RELAP5 volumes. Hence, the correct approach is to treat SCDAP heat structures the same as RELAP5 heat structures in the program structure and allow SCDAP to have the capability to automatically adjust the time step according to its own criteria.

Figure 4-5 repeats the SCDAP/RELAP5-3D© top level organization as discussed in Section 2. In implementing the time step repetition and time step control logic, the call to the SCDAP subcode has been split into two parts, SCDPRH (SCDAP Pre-HYDRO) and SCDPSH (SCDAP Post-HYDRO). The Pre-HYDRO portion of the code is the section of coding that has the capability of declaring the current time-advancement unsuccessful, and repeating the advancement with a smaller time step. Hence, the new calling sequence allows both SCDAP and RELAP5 to run at a smaller time step if the success criterion is not met in HYDRO.

SCDAP also has the capability of repeating a time step if subroutine SCDADV detects a current time step greater than the maximum time step allowed by the stability criterion based on the explicit coupling between radiation heat transfer from SCDAP heat structures and RELAP5 hydrodynamics. Various checks on solution acceptability are also used to control the time step. These include material Courant limit

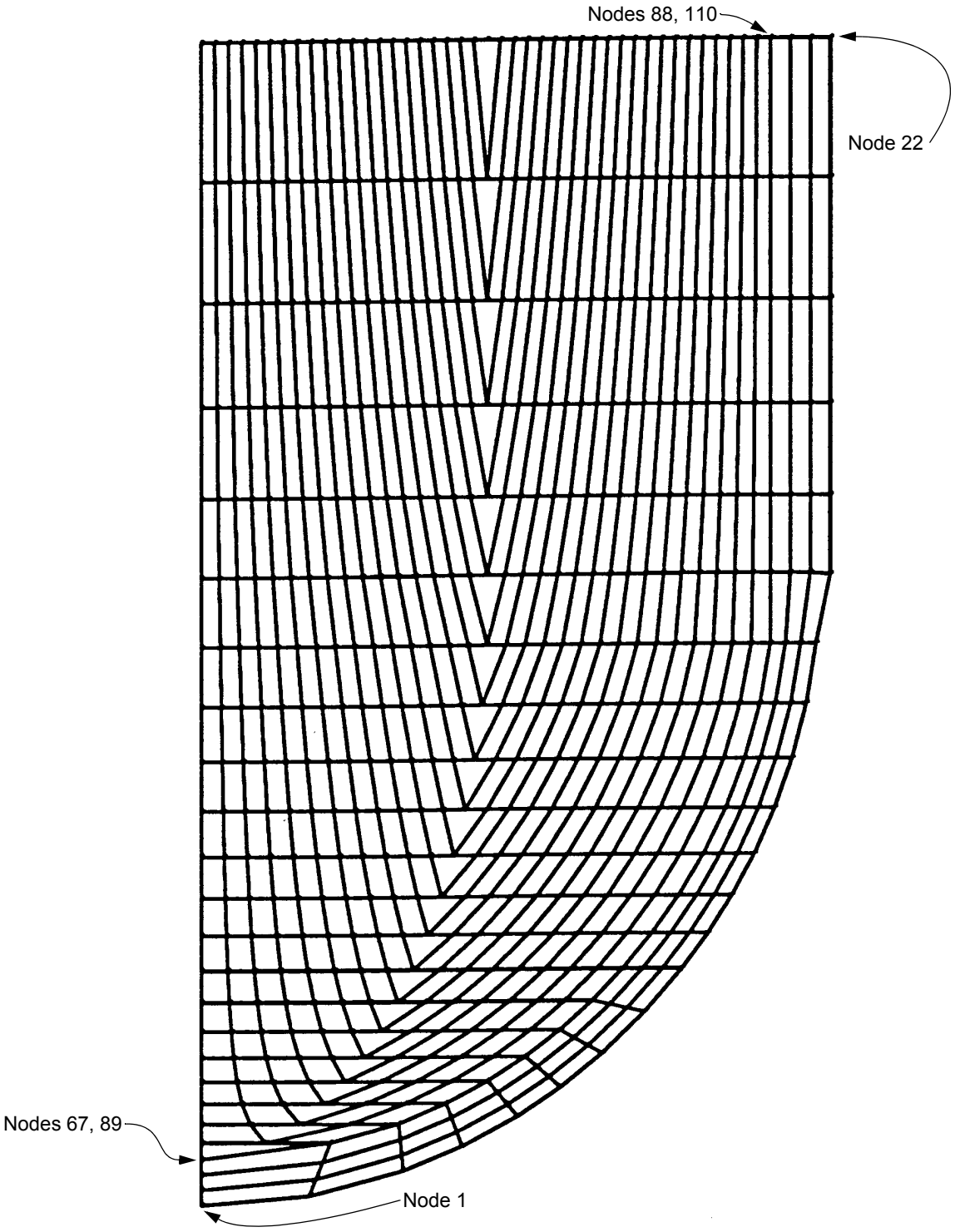

Figure 4-3. Finite element mesh representing the AP600 reactor vessel lower and lower head.

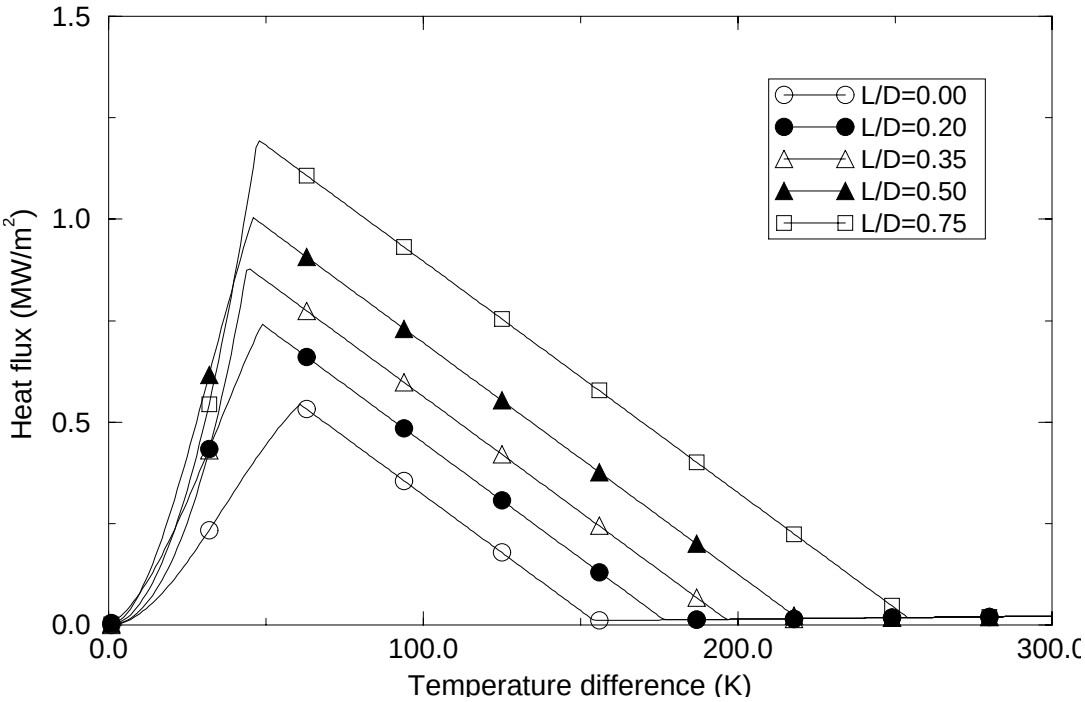

Figure 4-4. Predicted heat flux to a subcooled pool from lower head heat transfer correlations as a function of position and temperature difference.

checks, mass error checks, material properties out of defined ranges, water property errors, or excessive extrapolation of state properties in the meta-stable regimes.

The material Courant limit check is made before a hydrodynamic advancement takes place. The material Courant limit is evaluated for each hydrodynamic volume using the volume mass average velocity, i.e.,

$$(\Delta t_c)_i = \left[\Delta x \ \frac{\max(\alpha_f^n, \alpha_g^n)}{\max(|\alpha_f^n V_f^n|, |\alpha_g^n V_g^n|)}\right], i = 1, 2, \ldots, n \tag{4-26}$$

where n is the total number of volumes in the entire system.

The minimum of the Courant limit for each of the volumes is the Courant limit for the entire system.

The mass error check is made after the time step solution is nearly complete. Two types of mass error measures are computed. The first one checks the validity of density linearization and is defined as

$$E_m = \max\left[\left(\frac{|\rho_{mi} - \rho_i|}{\rho_i}\right), i\right] = 1, 2, \ldots, n \tag{4-27}$$

where ρ_{mi} is the total density of the i^{th} volume computed from the state relationship. The second one is a measure of overall system mass error and is given by

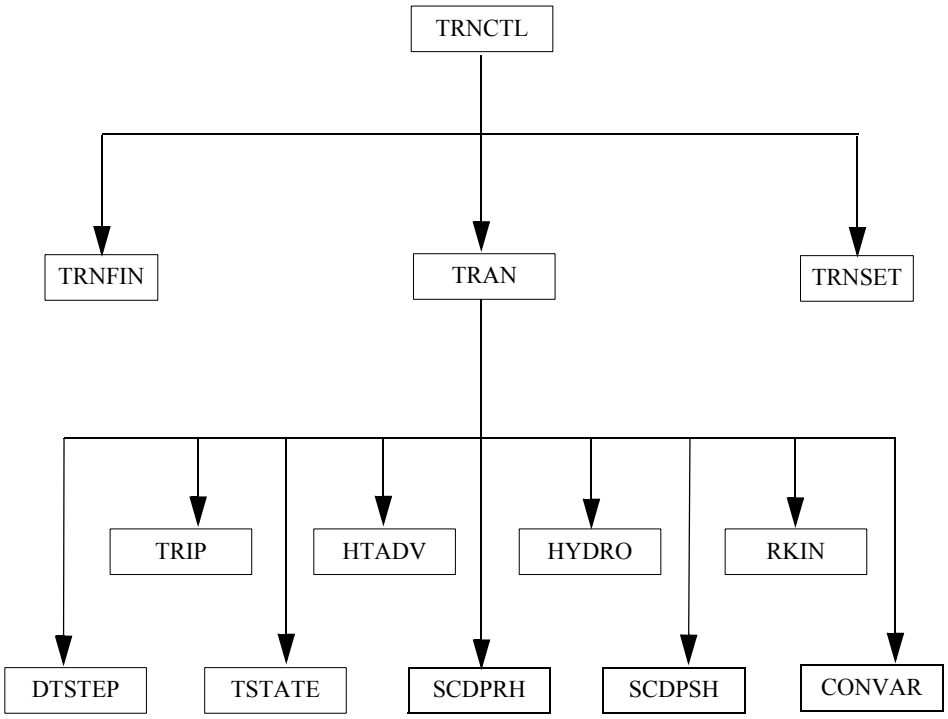

Figure 4-5. SCDAP/RELAP5-3D© top level organization.

$$E_{rms} = 2\frac{\sum_{i=1}^{N}[V_i(\rho_i - \rho_{mi})]^2}{\sum_{i=1}^{N}(V_i\rho_i)^2} \qquad (4\text{-}28)$$

where V_i is the volume of the i^{th} volume.

If either E_m or E_{rms} is > 0.008, the time step is rejected and repeated with one half of the time step size. Otherwise, the time step is accepted; and the next time step size is doubled if both E_m and E_{rms} are < 0.0008.

At any point in the solution flow, if a material property is found to lie outside the defined range, then the time step is halved and repeated. This process will proceed until the user-specified minimum time step is reached. If the minimum time step is reached without obtaining a valid solution, then the code calculation is terminated, and the last time step is repeated with a diagnostic dump printed. A program stop is encountered at completion of the step. This same procedure is applied for all property or extrapolation features.

4.4.2 Radiation Stability Limits

Experience has shown that an instability can occur in the hydrodynamic solution at time steps less than the Courant limit. As an example, the convective Courant limit, as determined by $\max[|v_g|, |v_f|]\frac{dt}{dx}$, could be 1.2 seconds. Hence, this problem can, in principle, be run with a time step of 0.5 seconds using the semi-implicit method without incurring stability problems. However, experience has shown that this is only true with radiation heat transfer disabled. This suggests that there is another stability limit in the vapor energy equation in the semi-implicit method. Ignoring the variations in the voids, temperatures, densities, and the interfacial heat transfer terms in the vapor energy equation, the finite-difference form of that equation is

$$U_j^{n+1} = \left(1 - a - \frac{h_r\ dt\ A}{mC_v}\right)U_j^n + aU_{j-1}^n \tag{4-29}$$

where

U_j^{n+1} = vapor internal energy at volume j at the current time step,

U_j^n = vapor internal energy at volume j at the previous time step,

U_{j-1}^n = vapor internal energy at volume j-1 at the previous time step,

a = $v_g\frac{dt}{dx}$,

A = surface area of the fuel rod interfacing the volume,

h_r = $\sigma\varepsilon(T_{w2}^2 + T_{g2}^2)(T_w + T_g)$,

m = mass of vapor at volume j (kg),

c_v = heat capacity of vapor (J/kg · K);

where

σ = Stefan-Boltzmann constant,

ε = emissivity of the vapor,

T_w = temperature of the fuel rod interfacing volume j,

T_g = temperature of the vapor at volume j.

In order for the difference scheme to be stable, the coefficients in the difference equation should be positive, implying that

$$1 - a - \frac{h_r\ dt\ A}{mC_v} > 0\ . \tag{4-30}$$

If the nearly-implicit scheme is used, then it can be shown that the scheme is stable if

$$dt < \frac{2mC_v}{h_r A} \quad . \tag{4-31}$$

On the other hand, if the semi-implicit scheme is used, the scheme is stable if

$$dt < \frac{1}{\frac{h_r A}{mC_v} + \frac{v}{dx}} \quad . \tag{4-32}$$

If the term $\frac{v}{dx}$ is small compared to the term $\frac{h_f A}{mC_v}$ in the inequality Equation (4-32), the radiation heat flux determines the limiting stability time step.

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